



Point Beach Nuclear Plant
6610 Nuclear Rd., Two Rivers, WI 54241

(920) 755-2321

NPL 99-0712

December 8, 1999

10 CFR 50.73

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, D.C. 20555

Ladies/Gentlemen:

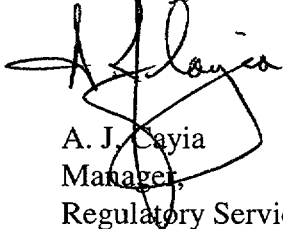
DOCKET NO. 50-266
LICENSEE EVENT REPORT 1999-013-00
INADVERTENT ESF ACTUATION
DURING POST MAINTENANCE TESTING
POINT BEACH NUCLEAR PLANT UNIT 1

Enclosed is Licensee Event Report 1999-013-00 for Point Beach Nuclear Plant Unit 1. This report is provided in accordance with 10 CFR 50.73(a)(2)(iv) as, "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF)." This report describes the inadvertent and unplanned actuation of an ESF signal during the post maintenance testing of a safety injection pump.

New commitments are identified in the corrective action section of this report by italics.

Please contact us if you require additional information.

Sincerely,



A. J. Cayia
Manager,
Regulatory Services & Licensing

Enclosure

CWK/tja

cc: NRC Resident Inspector
NRC Regional Administrator
NRC Project Manager

PSCW
INPO Support Services

IE22

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

FACILITY NAME (1) Point Beach Nuclear Plant, Unit 1	DOCKET NUMBER (2) 05000266	PAGE (3) 1 of 4
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TITLE (4)
Inadvertent ESF Actuation During Post Maintenance Testing

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
11	08	1999	1999	013	00	12	08	1999		05000
										05000

OPERATING MODE (9) N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)									
	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(viii)						
POWER LEVEL (10) 000	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(iii)	73.71						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 20.2203(a)(4)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	OTHER						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A						
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(vii)							

LICENSEE CONTACT FOR THIS LER (12)									
NAME Charles Wm. Krause, Senior Regulatory Compliance Engineer							TELEPHONE NUMBER (Include Area Code) (920) 755-6809		

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)										
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	

SUPPLEMENTAL REPORT EXPECTED (14)					EXPECTED SUBMISSION DATE (15)			MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	<input checked="" type="checkbox"/>	NO	<input type="checkbox"/>							

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On November 8, 1999, while completing post-maintenance testing of a safety injection (SI) pump, an inadvertent, unplanned Engineered Safety Feature (ESF) actuation occurred. The actuation of a safety injection signal occurred when a test jumper was installed to simulate an auto start signal for the SI pump motor. Point Beach Unit 1 was shutdown, cooled down and defueled at the time of this event. Systems and components required to operate as a result of the SI signal functioned as expected for the plant conditions. The cause of the event was a incorrect step in the PMT work plan. A NRC ENS notification was made at 1730 CST pursuant to 10 CFR 50.72(b)(2). The only safety significant result of this ESF was the isolation of service water to the spent fuel pool cooling heat exchangers. Service water was restored within approximately 50 minutes. The time to boil for the spent fuel pool at the time of this event was greater than 19 hours.

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TEXT CONTINUATION

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		1999	- 013	- 00	

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Event Description:

On November 8, 1999, at approximately 1600 CST, while completing post-maintenance testing (PMT) of the Point Beach Nuclear Plant (PBNP) Unit 1 "A" Safety Injection Pump (1P-15A), an inadvertent and unplanned Engineered Safety Features (ESF) actuation occurred. The PMT was scheduled to recover the 1P-15A pump after replacement of the pump motor control switch. The testing was being conducted by electrical maintenance personnel in accordance with an approved work plan. Step 26 of that plan directed the worker to install a test jumper which was intended to simulate an auto start signal for the 1P-15A pump motor. Instead the jumper installation resulted in a safety injection (SI) initiation signal.

At the time of this event, Unit 1 was in the 23rd day of a refueling shutdown with all fuel assemblies removed to the spent fuel pool. Based on a post event review, systems and components required to operate as a result of this SI initiation signal performed as expected. The following actions took place: (1) the "A" train Emergency Diesel Generators, G-01 and G-02, started (2) the P38A motor driven auxiliary feedwater pump started (3) non-essential service water loads, including the spent fuel pool heat exchangers, were isolated (4) a Unit 1 non-safety related motor control center supply breaker was tripped (5) a containment isolation signal was initiated. Because the unit was shutdown and de-fueled, those systems that would inject water into the reactor coolant system had been previously tagged out and no injection of ECCS water occurred. The plant operating crew entered procedures EOP 1.1, "SI Termination," AOP-18A, "Train 'A' Equipment Operation," and AOP-8F, "Loss of Spent Fuel Pool Cooling," and recovered from this event.

The NRC was notified of this ESF actuation by an ENS telephone call at 1730 CST on November 8, 1999.

Cause:

The direct cause of this event was a technically incorrect work plan. Although this SI pump motor control switch replacement had previously been completed on the PBNP Unit 2 SI system, the work plan for the Unit 1 PMT was prepared by a different planner and did not include the same sequence of steps as the Unit 2 work plans. The review and approval process for the Unit 1 work plan did not receive an in-depth review by a technically knowledgeable reviewer and the work plan deficiency was not caught. A root cause evaluation to determine whether program or process related issues contributed to this mistake will be completed.

Corrective Actions:

1. A team comprised of a senior electrical maintenance craft worker, a senior engineer from system engineering, and a planner performed an in-depth review of the work plan and drawings. Their evaluation confirmed that the existing work plan was technically incorrect and, given the conditions created by that work plan, the SI signal actuation was the expected system response. The team created a work plan addendum to perform the PMT. Following appropriate reviews and approvals, the revised work plan was then executed and the PMT performed without incident.
2. A formal Root Cause Evaluation is being performed to identify the reasons why the work plan was not adequately developed and to identify why reviews and a walk-down of the work plan did not uncover the deficient steps. Corrective actions identified

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in this RCE will be entered into the licensee's corrective action program and tracked to completion using established processes.

Component and System Description:

The Engineered Safety Features Actuation System (ESFAS) is described in Section 7.3 of the PBNP FSAR. The ESFAS detects plant conditions that require automatic ESF equipment operation, and actuates the appropriate ESF equipment when preset limits are reached. ESFAS subsystems monitor plant parameters indicative of different accidents. When the minimum number of channels of a monitored variable reach a preset limit, trip bi-stables satisfy coincidence logic for an individual subsystem and the subsystem is automatically initiated. ESFAS subsystems include: Safety Injection Actuation, Containment Isolation, Containment Ventilation Isolation, Containment Spray Actuation, Steam Line Isolation, Auxiliary Feedwater Pump Start, and Feedwater Isolation.

A manual or automatic safety injection signal initiates: 1. High head safety injection and low head (RHR) pump start and valve stroking 2. Emergency diesel generator start 3. ESF (safeguards) load sequencing 4. Reactor trip 5. Motor-driven auxiliary feedwater pump start 6. Service water pumps start and non-essential branch isolation (if required) 7. Containment fan cooler start and increased fan cooling water flow 8. Containment isolation of nonessential systems (from automatic SI signal only) 9. Containment ventilation isolation 10. Feedwater isolation 11. Permissive for Steam Line Isolation.

Safety Assessment:

At the time of this inadvertent ESF actuation, the unit was shutdown with all fuel assemblies removed to the spent fuel pool (SFP). Given the plant status and equipment conditions at this time, the plant response to the ESF/SI actuation signal was as expected. There were no system, structure or component failures or abnormal responses observed. The only plant response to the ESF/SI signal of potential safety significance was the automatic isolation of service water to the SFP cooling system heat exchangers. At the time of this event, the status of the SFP cooling system was being monitored as directed by NP 10.3.6, "Outage Safety Review and Safety Assessment." SFP cooling conditions were being evaluated and documented by the Shift Technical Advisor on a once per 12 hour shift schedule using the "PBNP Shutdown Safety Assessment and Fire Condition Checklist." The average spent fuel pool water temperature at the time of this event was approximately 84°F. The time to boil for the SFP was conservatively calculated as greater than 19 hours. Both these parameters were being monitored and recorded on the PBNP Shutdown Safety Assessment Checklist, which was being updated twice per day at this time. Service water to the SFP was restored manually in accordance with AOP 8F approximately 50 minutes after the SI signal had stroked shut the service water supply valves to the SFP heat exchangers. The average SFP water temperature was observed to have increased slightly to approximately 85°F. Based on the plant conditions and the response of the crew to promptly restore SFP cooling, the safety significance of this event was minimal.

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System and Component Identifiers:

The Energy Industry Identification System component function identifier for each component/system referred to in this report are as follows:

<u>Component/System</u>	<u>Identifier</u>
Reactor Coolant System	AB
Auxiliary Feedwater System	BA
Safety Injection System	BQ
RHR Low Pressure SI System	BP
ESF Actuation System	JE
Containment Isolation Control System	JM
Fuel Pool Cooling System	DA
Switch, Power	JS
Pump	P
Heat exchanger	HX

Similar Occurrences:

A review of recent LERs (past three years) identified the following event reports which involved inadvertent actuation of ESF systems or equipment:

<u>LER NUMBER</u>	<u>Title</u>
266/98-024-00	Inadvertent Emergency Diesel Generator Start
266/98-014-00	Emergency Safety Feature Actuation Automatic Start Of A Service Water Pump
266/98-006-00	Unanticipated Partial Service Water System Isolation During A Special Test
266/98-002-00	Failure Of The High Voltage Station Auxiliary Transformer
266/97-034-00	Unplanned Loss of Voltage on Train B Safeguards Buses